NASA TECHNICAL TRANSLATION NASA TT F-11,705

NASA TT F-11,705

THE SPHERICAL PILE REACTOR AS THERMAL THORIUM BREEDER
I. GRAPHITE AS MODERATOR

S. Brandes

Translation of: "Der Kugelhaufenreaktor als thermischer Thioriumbruter, I. Graphit als Moderator" Report of the Julich Nuclear Research Installation No. 474, Institute for Reactor Development Jul-474-RG Institute for Reactor Development, Julich Nuclear Research Installation, Nordrhein-Westphalia State, Incorporated Association, Julich, Federal Republic of Germany, January 1967

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION
WASHINGTON, D. C. 20546 MAY 1968

TABLE OF CONTENTS

LIST OF SYMBOLS

A	Number of absorbed neutrons per neutron absorbed in the fuel
B^2	Buckling (cm^{-2})
C	Breeding factor
L	Number of neutrons lost by leakage per neutron absorbed in the fuel
Q	Average power density (MW/m^3)
R	Specific power ($kW_{th}/g_{[fissionable\ fuel]}$)
S	Moderation ratio (moderator/fissionable fuel)
T_B	Radiation time of the fuel elements in the reactor (a)
T_K	Cooling time of the fuel after removal from the reactor (a)
T_D	Doubling time (a)
V	Reprocessing losses (%)
ϕ	Neutron flux ($cm^{-2}\ sec^{-1}$)
σ	Microscopic effective cross-section (b)
η	Number of liberated neutrons per neutron absorbed in the fuel

NASA TT F-11,705

The Spherical Pile Reactor as Thermal Thorium Breeder
I. Graphite as Moderator*

S. Brandes

ABSTRACT. The breeder characteristics of a graphite-moderated high temperature reactor of the spherical pile type for the thorium 232-uranium 233 breeder cycle are studied, investigation being conducted from the standpoint of minimum doubling times. Economic considerations are ignored. The BABS program (a burn-up program for a thorium converter developed by the DRAGON project), expanded and improved for the design of breeder reactors (GABS program) was employed for calculations.

I. Introduction

The possibility of generating fissionable fuels from thorium 232 or uranium 238 by neutron capture with subsequent conversion of the atom nucleus is utilized in breeder reactors. Thus a neutron capture in uranium 238 leads to the formation of a plutonium 239 nucleus which is highly subject to fission by fast neutrons while a neutron capture in thorium 232 finally leads to the formation of a uranium 233 nucleus. U^{233} is a good fissionable fuel for thermal reactors (see Fig. 1). /3**

With the simultaneous presence of breeder material (Th^{232} or U^{238}) and fissionable fuel in the reactor it is now possible to maintain a chain reaction. Secondly, the burned-up fissionable fuel is largely replaced by "bred" fissionable fuel. With a suitable design of the fuel circuit, such a system is not only incapable of making do with a single fissionable fuel

* All calculations in this study were carried out on the IBM-7090 of the University of Bonn.
** Numbers in the margin indicate pagination in the foreign text.

/4

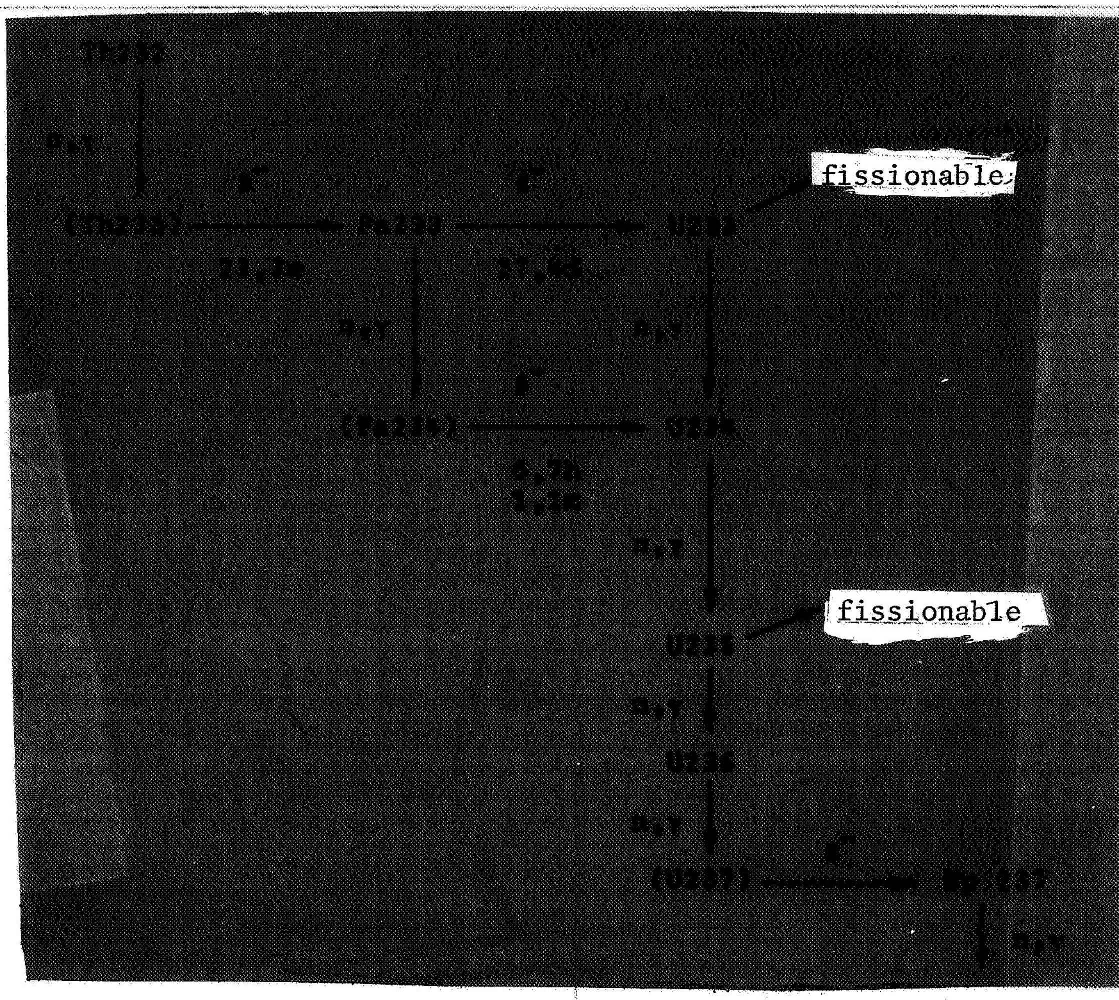


Figure 1. Schematic Drawing of Conversion in the Case of the Th^{232} - U^{233} Breeding Cycle (The Nuclides in Parentheses Were Not Considered in the Calculation of Burn-up.

charge, but it can generate more fissionable fuel than it consumes. If the latter is the case, so much excess fissionable fuel is generated after the doubling time that an initial charge is available for another reactor of this type.

Since such breeder reactors require only one fuel feed to relatively abundant U^{238} and Th^{232}, its development is necessary due to the shortage of U^{235} fissionable fuel. Breeder reactors also need no more expensive enrichment systems. Admittedly, a reprocessing system is necessary for the fuel.

In the present study, we investigate the breeder characteristics of a graphite-moderated high temperature reactor of the spherical pile type for the thorium 232 - uranium 233 breeder cycle.

* Commas in the figure indicate decimal points.

The spherical pile reactor is well-suited as a thorium breeder because of its continuous charging and discharging (no control rods, no excess reactivity), because of its small neutron leakage losses (large reactor core) and because strongly absorbing structural material is not used (only graphite). /5

The studies were conducted with regard to minimum doubling times.

Economic considerations were disregarded.

The BABS program served as a basis for the calculations, a burn-up program for a thorium converter developed by the DRAGON project with continuous loading and two types of fuel elements (feed and breed element).

This program was expanded and improved for the calculation of breeder reactors (GABS program).

2. The Fuel Circuit of the Thorium Breeder

In the study conducted here, we proceeded from the condition of equilibrium of the reactor core, i.e., the start-up condition up to the attainment of the equilibrium state in which continuous loading was not taken into account.

Thorium as well as bred, reprocessed uranium was used as fuel in the reactor core in the form of changed-off particles in spherical graphite combustion elements of 6 cm diameter.

The changed-off particles are small fuel particles up to 1 mm diameter which are charged with pyrolitic carbon and thus reach a very high burn-up rate without damage (up to 250,000 Mw d/t at 1,600°C).

The feed of the combustion elements in the reactor occurs continuously with simultaneous removal of the fuel elements having the highest degree of burn-up. In this way, the reactor core volume is kept constant. /6

This loading is controlled by continuous rotation of the spheres. The spheres are removed individually from the floor of the reactor core, tested for damage and subjected to testing until completion of burn-up. Acceptable spheres are put back into the reactor core from above. The continuous rotation of the spheres has the advantage that all combustion elements are burned up almost uniformly.

After radiation in the reactor, the fuel is separated from the fission products in the reprocessing installation. The present water reprocessing takes about 0.8 year, including the precooling time for the decay of the high radioactivity of the fuel as well as the fabrication time and transportation times. In addition, about 0.5% uranium loss must be reckoned with during the reprocessing. The fission products can be almost completely removed.

The long reprocessing period has a very unfavorable effect on the doubling time. In addition, the water reprocessing is very expensive. Thus, a method has been proposed by SCHULTEN which is particularly suited for the reprocessing of changed-off particles. The fission products are vaporized at about 2,000°C-2,200°C. The developmental goal of this regeneration method is to achieve a 10% reduction of the fission products from its initial value already after a few hours [3]. But neptunium, zirconium, molybdenum and technetium isotopes are exceptions which cannot be removed. To prevent the concentration of these nuclides from being too high, we are forced to include a water reprocessing after several reprocessing operations with the regeneration method.

After reprocessing of the fuel, the excess part of the uranium isotopes is separated and stored. The remaining fuel is again fed to the reactor with /7
additional fresh thorium.

The condition of equilibrium is studied which is present when the amount of uranium reduced by reprocessing losses and by breeding gain is fed repeatedly to the reactor core. All uranium isotopes then have almost their saturation concentration in the fuel.

If the reactor core consists of only one fuel zone, then the neutron outflow is above 2.5% (see Diagram 1) even at 2,000 MW_{el} at usual power densities. Thus it is advisable to consider a blanket to reduce the neutron leak losses from the reactor, i.e. a thorium zone in the reactor between the actual reactor core and the reflector. This thorium zone is formed in such a way that the thorium atoms uniformly distributed in a homogeneous reactor core are concentrated in certain areas in a reactor with a blanket, i.e. they are concentrated in the blanket and in an interior zone (flux reduction!). The neutron outflow can be arbitrarily reduced according to the thickness of the outer thorium zone.

3. Breeding Factor, Doubling Time, Breeding Gain and Breeding Component

The breeding factor is defined as the ratio of fissionable fuel gain to fissionable fuel loss. If this definition is applied to the reactor core, we obtain the "breeding factor" of the reactor core. The protactinium is given an opportunity to convert to U^{233} by means of β^- decomposition by the cooling time determined by reprocessing of the fuel. To a certain extent, protactinium is a latent fissionable fuel carrier which increases the gain of fissionable fuel during the cooling time. Since a reprocessing of the fuel always takes place in breeding reactors, and generally, since there is a cooling time (in the case of the regeneration method, the reprocessing /8
period is so short that we can no longer speak of a cooling time), the breeding factor C must be determined in the following familiar way:

[redacted] (4.01)

where η = the number of liberated neutrons per neutron absorbed in the fuel

A = number of absorbed neutrons per neutron absorbed in the fuel

L = number of neutrons lost by leakage per neutron absorbed in the fuel.

The breeding factors given in this study have all been determined in this manner.

The η value which occurs in the breeding factor formula, in addition to the fissions in U^{233} and U^{235}, also includes the rapid fissions in thorium, protactinium and U^{234} as well as U^{236} and in the necessary form is a group value of the total neutron flux spectrum. Characteristically calculated η-values of this type are listed in Table 1.

TABLE 1. η- VALUES OF A GRAPHITE MODERATED SPHERICAL PILE REACTOR BEING USED AS A THORIUM BREEDER AS A FUNCTION OF THE MODERATION RATIO$^{+)}$ S. THE LOWER VALUES CORRESPOND TO A POWER DENSITY OF 7 W/cm^3, THE UPPER-ONES TO A POWER DENSITY OF 4 W/cm^3 (NEUTRON TEMPERATURE 900°K). /9

[illegible] **

With regard to the data in Table 1 it should be noted that the rapid fissions component in the η-value amounts to about 0.5%. The number of neutrons lost by leakage is taken into account in the burn-up calculation by the introduction of a buckling term B^2. A reactor with a thermal power of about 4,650 MW_{th} (2,000 MW_{el}) is taken as a basis for the studies. The B^2 terms corresponding to the individual power densities of the reactor core are listed in Table 2. A distinction is made between two reactor cores with different blankets (blanket A and blanket B). Blanket A reduces the neutron outflow by about 50% comparison to a single zone reactor, blanket B by 75%.

---

*By the moderation ratio we mean the ratio of the number of moderator atoms (here graphite) to the number of fissionable fuel atoms.

** Commas indicate decimal points.

TABLE 2. BUCKLING TERM B^2 TO DESCRIBE NEUTRON LEAKAGE FOR A 2000 MW_{el} SPHERICAL-PILE REACTOR AS A FUNCTION OF POWER DENSITY. /10

Power Density $\frac{MW}{m^3}$		[redacted]
$B^2 \cdot 10^5$ cm^{-2} (The numbers in brackets correspond to that part of neutrons lost by outflow with respect to the total neutron loss by absorption and leakage at S = 4,000).	With Blanket A	
	With Blanket B	

The neutron losses by leakage and absorption in the graphite with respect to a neutron absorbed in the fuel are listed in Table 3 (i.e. $L + A_{graphite}$ of Eq. 4.01).

The xenon portion ($A_{Xe^{135}}$) increases with increasing power density and increasing moderation ratio:

from 0.026...0.033 (for 4...7 MW/m^3) at S = 3,000.

from 0.036...0.040 (for 4...7 MW/m^3) at S = 5,000.

The Sm-portion ($A_{Sm^{149}}$) depends on the level of specific power and on the burn-up (0.0070 to 0.0075 at 5,000 MWd/t; 0.0080 to 0.0085 at 10,000 MW d/t; 0.0090 to 0.0095 at 20,000 MWd/t; low values for 0.6 kW_{th}/g, high values for 1.7 $kW_{th}/g_{[fissionable\ fuel]}$. It will be noted that values averaged over time are significant for the reactor core).

Assuming that no protactinium undergoes fission in the cooling times or, /12
when it does undergo fission, the fuel has been burned-up for a long time (i.e. equilibrium concentration of Pa^{233} in the reactor core), the Pa-portion ($A_{Pa^{233}}$) at most and has the following values: 0.030....0.050 (for 4...7 MW/m^3) at S = 3,000 and 0.038...0.060 (for 4...7 MW/m^3) at S = 5,000. (Pa counts double in the balance sheet). The portion of U^{236} is also of interest on the balance sheet. It is actually at its maximum when the Pa-portion is at its maximum. i.e. Table 4 is applicable assuming that protactinium does not

TABLE 3. NEUTRON LOSSES THROUGH LEAKAGE AND GRAPHITE ABSORPTION FOR A 2,000 MW_{el} SPHERICAL-PILE REACTOR AS A FUNCTION OF THE MODERATION RATIO AND POWER DENSITY.

Moderation Ratio S		2000					4000					8000				
Average Power Density MW/m^3		[illegible]	[illegible]	[illegible]	[illegible]	7	3	4	5	6	7	3	4	5	6	7
Neutron Losses by Leakage & Graphite Absorption with Respect to a Neutron Absorbed in the Fuel	a)	[illegible]	[illegible]	[illegible]	0,048	-	0,043	0,048	0,050	0,055	-	0,038	0,044	[illegible]	[illegible]	-
	b)	-	[illegible]	[illegible]	0,030	0,032	-	0,033	0,035	0,027	0,038	-	0,040	0,042	[illegible]	[illegible]

Key: a) With Blanket A; b) With Blanket B.

* Commas indicate decimal points.

undergo fission during long cooling times.

TABLE 4. NEUTRON LOSSES BY ABSORPTION IN U^{236} WITH RESPECT TO A NEUTRON ABSORBED IN THE FUEL.

Moderation ratio	[redacted]
Power density MW/m^3	[redacted]
Burn-up in MW d/t	[redacted]

The remainder of the A terms represent the neptunium and the remaining fission products. Assuming very short cooling times, the remaining term amounts to 0.012 at 5,000 MWd/t, 0.023 at 10,000 MWd/t and 0.043 at 20,000 MWd/t.

The breeder factor C and the specific power R ($kW_{th}/g_{[fissionable\ fuel]}$ /13
are the determining factors of the doubling time T_D.

If the excess $(C-1)\ \Sigma^a\phi - V\overline{N}/T_B$ ($\Sigma^a\phi$ = absorption rate in the fissionable fuel, V = reprocessing losses, $\overline{N}$ = the number of fissionable fuel atoms per unit of volume of the reactor core, T_B = radiation time in the reactor) are collected, then after T_D years we have the required amount for the first inventory of another reactor:

[redacted] (4.02)

In general, this amount is not yet sufficient for operation because another fuel is needed in the ratio cooling time T_K to radiation time T_B in order to fill up the fuel circuit.

This formula for the doubling time can be explained more completely as follows:

[redacted] (4.03a)

[redacted] (4.03b)

The last equation can be represented somewhat more clearly for rough calculations by the introduction of specific power R.

[redacted] (4.04)

The derived formulas for the doubling time are valid for the first breeder reactor. If the installed power in the breeder reactors has become so great that the doubling gain can immediately be used again, then the amount of fissionable fuel increases organically. The doubling times according to equation (4.03b) are then reduced by the factor ln2. The doubling times given in this study are understood for the first breeder reactor. To attain breeding times in the case of organic growth, the values must actually be multiplied by ln2. /14

If the breeding factor is greater than one, then a breeding gain results. This breeding gain is defined as the amount of uranium in $g/MW_{th}.a$, which is separated after reprocessing as excess fuel and is no longer needed for the continuous operation of the reactor.

If the breeding gain is related to the yearly throughput of uranium through the reprocessing installation, then we obtain the breeding component (in %). This quantity will be zero when the breeding gain becomes zero, and also when at a very short radiation time the throughput (in the case of finite breeding gain!) becomes very large. Thus there are radiation times in which the breeding component has a maximum. That is interesting because a reprocessing of the fuel always involves material losses (in per cent with respect to throughput). To make these as insignificant as possible, the radiation times must be selected with a maximum breeding component.

With the aid of the breeding gain, it is simple to determine the doubling time:

[redacted] $\frac{\text{Fuel charge}}{\text{Breeding gain}}$ [redacted] (4.05)

4. Discussion of the Computer Results /15

4.1. Discussion of the Influence of Cooling Time and Reprocessing Losses on the Breeding Characteristics

Before the fuel is reprocessed, the water reprocessing methods require a certain cooling time. During this time the protactinium 233, among other things, has an opportunity to undergo fission.

If the reprocessed fuel is put back into the reactor, the breeding rate is higher because of the reduction or even the absence of the absorbing protactinium. This applies until the saturation concentration has built back up again.

The breeding rates are given in Diagram 2 and in Diagram 3 the breeding gain and the breeder component are presented as a function of the cooling time over the radiation time.

The favorable influence of the cooling time is lost with increasing radiation time. A cooling time of 0.5 years is equivalent to an infinitely long cooling time.

Since the cooling time is extended into the doubling time, the longest logical cooling time with respect to the doubling times amounts to 0.25 years. The water reprocessing methods which take at least 0.8 years, are thus not very advantageous for a thorium breeder.

The influence of the reprocessing losses on breeding gain is explained in Diagram 4. In order to obtain the breeding component at certain reprocessing losses, the breeding component at 0% reprocessing losses is reduced by the value of the reprocessing losses (e.g. 0.4%). From the curve thus obtained we read off that in the presence of reprocessing losses to achieve an effective breeding gain there is a minimum and a maximum radiation period. At both radiation periods the breeding gain is zero and the doubling time is infinite. Correspondingly, there is a maximum breeding gain at a certain optimum radiation time. This breeding gain is increased by the uranium 233 (3-6%) resulting from the protactinium undergoing fission and is inversely proportional to the
doubling time. The curve of the doubling time as a function of radiation time /16
is plotted in Diagram 5 for purposes of interpretation. The minima of the doubling times resulting from the reprocessing losses are characteristic. In the following, the conditions of interpretation which lead to minimum doubling times are of interest.

4.2. Breeding Characteristics of a Thorium High Temperature Reactor With a Blanket

The following values for the reactor core were determined for the investigations in this report:

Thermal power	4,650 MW
Moderator temperature	900°K
Fuel temperature	950°K
Number of graphite atoms	$5.1 \cdot 10^{22}$ cm^{-3}
Effective cross-section of absorption for graphite	4.0 mb

(2,200 m/s)

Self-shielding factor for TH 232
(when not otherwise indicated) 1.0

After the water reprocessing there should be only insignificant amounts of fission products in the fuel.

For a graphite-moderated thorium reactor with blanket A, studies for average power densities of 3 - 6 w/cm^3, moderation ratios from 3,000-5,000 and cooling times of the fuel were conducted during reprocessing from 0 to 0.25 years.

Optimum cooling times (Diagram No. 6), optimum moderation ratios (Diagram No. 7) and optimum radiation times (Diagram No. 8) can be given for /17 the given reprocessing losses which lead to minimum doubling times (Diagram No. 12). In Diagrams 10 and 11 the FIFA-values and burn-ups corresponding to the optimum radiation times are plotted. Diagram No. 9 contains the optimum breeding factors.

The doubling time decreases with decreasing power densities (large reactor core, small leakage losses) and small reprocessing losses. But the attainable doubling times are too high to be of interest for a breeder. In addition, the specific power is less than 1 $kW_{th}/g_{(fissionable\ fuel)}$. The burn-ups are in the order of magnitude of 3,000 - 7,000 MWd/t and are too small to achieve economical fuel circuit costs in the foreseeable future even with consideration of the reduction of reprocessing and fabrication costs. Since the doubling times are too high from the beginning, it is logical to study the case of a reactor which operates with infinite doubling time. The breeding factor is somewhat higher than one to equalize the reprocessing losses. Such a reactor requires only an initial core supply of fissionable fuel.

The burn-ups which can be achieved with such a system are plotted in Diagram 13.

The ratios are somewhat more favorable with blanket B which in comparison to blanket A reduces the neutron leakage losses by one half.

Optimum conditions which lead to minimum doubling times are presented for this case in Diagrams 14-18. The minimum doubling time which can be achieved is found in Diagram No. 19. By comparing Diagrams No. 7 and 16, it can be determined that the accepted blanket B allows a 30% higher specific power than blanket A.

To obtain lower doubling times, it is necessary that:

1. the average power density be small (<4 MW/m^3),

2. the reprocessing losses be small (<0.5%, or better < 0.2%), /18

3. the cooling time be shorter than 0.1a,

4. the radiation time be less than 0.4 years corresponding to burn-ups < 5,000 MWd/t.

The doubling times which can be achieved for the first breeder reactor (specific power 1.2 - 1.3 kW_{th}/g) are 40-60 years; that corresponds to a doubling time of about 28-40 years if the organic growth of the fissionable fuel is guaranteed by a sufficiently large number of thorium breeders.

The above-named conditions cannot yet be fulfilled at present with the usual water reprocessing method (long cooling time and large losses). Thus Diagram 20 shows the doubling times which can be achieved with the use of the water reprocessing method. They are much less favorable. Diagram 20 shows that a graphite moderated reactor with a strictly dimensioned blanket does not achieve a logical doubling time with the use of the water reprocessing method. Thus it is more economical to prolong the radiation time and thus the burn-up to such an extent that the doubling time becomes infinite.

Diagram No. 22 shows such a maximum burn-up.

The influence of the geometric self-shielding is studied in Diagram No. 21. Since it is probable that the self-shielding factor in the case of the combustion elements in question is not less than 0.9, this effect can be neglected.

4.3. Influence of the Regeneration Method on the Doubling Times of a Graphite Moderated Thorium Breeder With Blanket /19

The long cooling time necessary in the case of water reprocessing has an especially unfavorable effect on the doubling time. The regeneration method does not have this disadvantage. Indeed the non-diffusable fission products of the breeding factor are steadily reduced by the slow build-up. In general, it can be said that the breeding factor declines by 1% yearly. One is forced to reprocess water after 2.5 - 3 years. The following assumptions were made for a study of the influence of the regeneration method on doubling time:

Arbitrary cooling time in the case of regeneration (i.e. from 0 to 0.2 years),

0% uranium losses (i.e. < 0.5 $^0/_{00}$) in the case of regeneration,

build-up of Np, Tc, Mo, Zr (no precipitation in the case of regeneration),

Residual fission products were reduced by 90% in their concentration by regeneration.

$5^{0}/00$ uranium losses in the case of water reprocessing,

0.8 a cooling time in the case of water reprocessing,

Assuming a B blanket.

The results of the preceding chapter show that the optimum radiation time must also be short with a short cooling period. Thus, the studies were limited to radiation times of 0.25 and 0.5 years.

Because of the long cooling time in the case of water reprocessing, it will be advantageous to conduct many regenerations before beginning water reprocessing. (See Diagram No. 23). After a certain number of regenerations, the reduction of the breeding factor by the build-up of the above-mentioned non-volatile nuclides outweighs the gain in doubling time due to which the extremely short cooling time appears in the case of the regeneration. /20

Since the cooling time of 0.8 years in the case of water reprocessing is longer than the radiation time (0.25 or 0.5a), in addition to the actual fuel charge in the operation of the reactor, a second must be available in order to continue reactor operation after the start of water reprocessing.

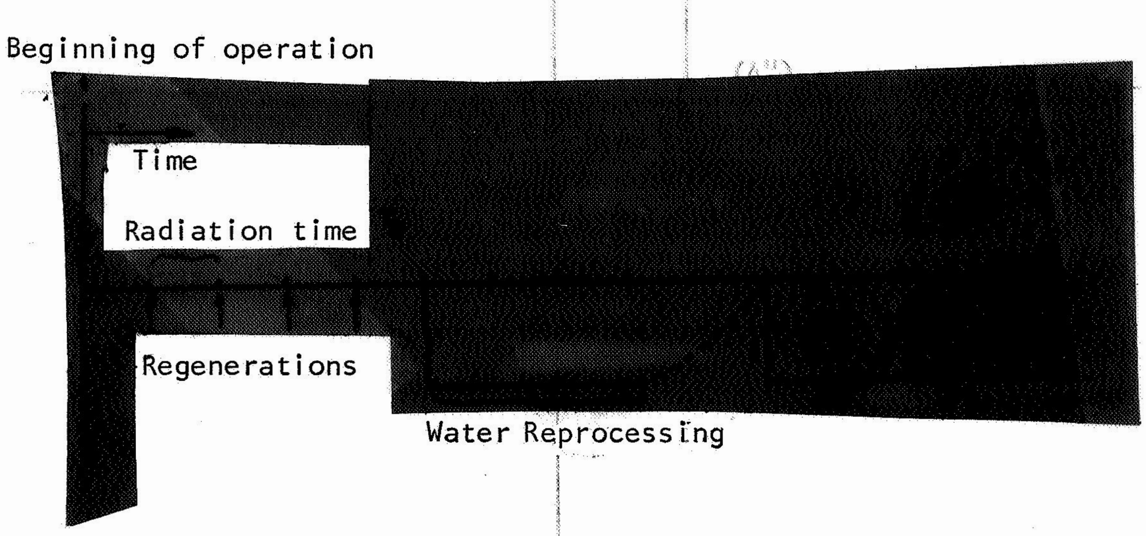


Figure 2. Distribution of the Times for Water Reprocessing During the Operating Life of the Reactor.

Thus temporarily, a complete fuel charge of the reactor core is present in the water reprocessing (see Fig. 2).

After the water reprocessing, the fuel charge lies unused until it is used again in the reactor at the beginning of reprocessing of the next reactor charge.

If the number of regenerations between the water reprocessing is so great that in addition to the required time for one water reprocessing there is also time for a second, a second reactor can be charged with the unused fuel charge whose fuel is now reprocessed and subsequently will be again available to the first reactor. (see Fig. 3) /21

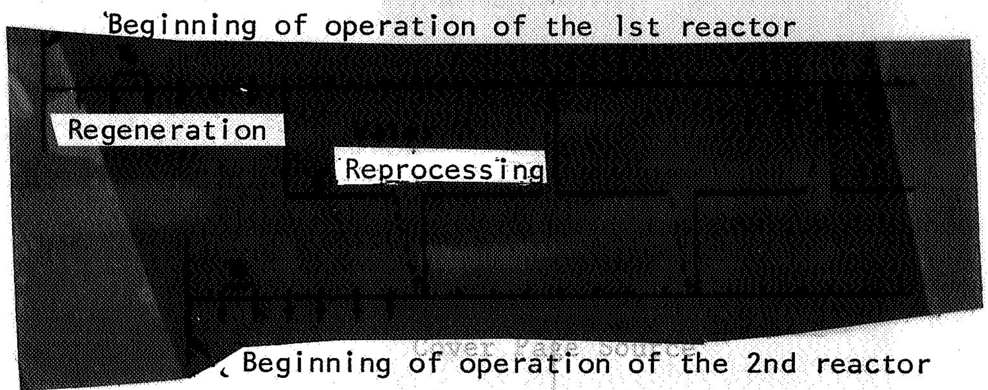


Figure 3. Reactor Operation With Two Reactors and Three Fuel Core Charges (Regeneration and Water Reprocessing.

In this manner, two reactors now operate with three fuel core charges.

If the number of regenerations between the water reprocessings is further increased, then still more reactors can be included in this manner. Finally, the method can be simplified to such a degree that water reprocessing and the regeneration method can be applied simultaneously. But this case will not be studied further here. If the number of reactors is increased which operate with this method, then the doubling time is reduced noticeably (see Diagram 23). With an increasing number of reactors, however, the gain in doubling time becomes increasingly smaller so that after a certain time the influence of the breeder factor breeding factor which becomes steadily worse with an increasing number of regenerations prevails.

From Diagram No. 23 we read that water reprocessing is necessary after about 2.5 years. For the analysis of the calculations with regard to minimum doubling times, seven to ten regenerations were assumed at a radiation time of 0.25, five to seven regenerations were assumed at a radiation time of 0.5 before a water reprocessing. /22

The favorable interpretation data are shown in the following tables:

TABLE 5. MINUMUM DOUBLING TIME WITH THE USE OF THE REGENERATION METHOD IN THE CASE OF A GRAPHITE-MODERATED THORIUM BREEDER WITH BLANKET A.

Average Power Density Q MW/m^3	Radiation Time T_B [a]	Number of Regenerations Before a Water Reprocessing	Optimum Cooling Time at Regeneration T_K [a]	Optimum Moderation Ratio S_{opt}	Minimum *Doubling Time* T_D [a]	Number of Reactors Involved
4	0,25	7	0	5000	68	[illegible]
		10	0	5000	63	[illegible]
	0,5	5	0	4800	76	[illegible]
		7	0	4800	87	[illegible]
6	0,25	7	0,1	5000	88	[illegible]
		10	0,1	5000	83	[illegible]
	0,5	5	0,1	4000	119	[illegible]
		7	0,1	4000	136	[illegible]

5. Final Considerations /24

The minimum doubling times of the first reactor as a function of various reprocessing methods are shown in Diagram No. 24 for a thorium breeder with the assumption that there is a blanket which reduces the neutron outflow by one-fourth in comparison to a single zone reactor. With the regeneration method, ten regenerations were carried out before a water reprocessing took place, at a radiation time of the fuel elements in the reactor of 0.25 years. The regeneration of the fuel has a very advantageous effect on the duplication times as opposed to the single water reprocessing.

The graphite moderated-thorium high-temperature reactor can be logically used as a genuine breeder. It is possible to aim at only an infinite doubling time. The possible burn-up then achieves an order of magnitude of 20,000 to 30,000 MW d/t when water reprocessing of the fuel is assumed.

REFERENCES

1. Blomstrand, J. H.; "A Method for Analysis of Equilibrium Fuel Cycles", (BASS and BABS) D. P. Report to be issued.
2. Nyffenegger, U.: "The Dragon Nuclear Data Library Three", D.P. 420, May 1966.
3. Hecker, R.; "Developmental Problems of a High Temperature Thorium Breeder Reactor", Jul-382-RG, Sept. 1966.
4. Brandes, S.: "Study of the Breeder Characteristics of a Graphite Moderated Pebble Bed Reactor in Use as a Thorium Breeder", IRG-66-38, Oct. 1966.
5. Brandes, S.; "Description of the Burn-up"; GABS, IRG-report in preparation.

Translated for the National Aeronautics and Space Administration under Contract No. NASw-1695 by Techtran Corporation, P. O. Box 729, Glen Burnie, Maryland, 21061.

Diagram No. 1 Buckling and Neutron Outflow of a Graphite Moderated Thorium Breeder
(η = 43%, Single-Zone Core)

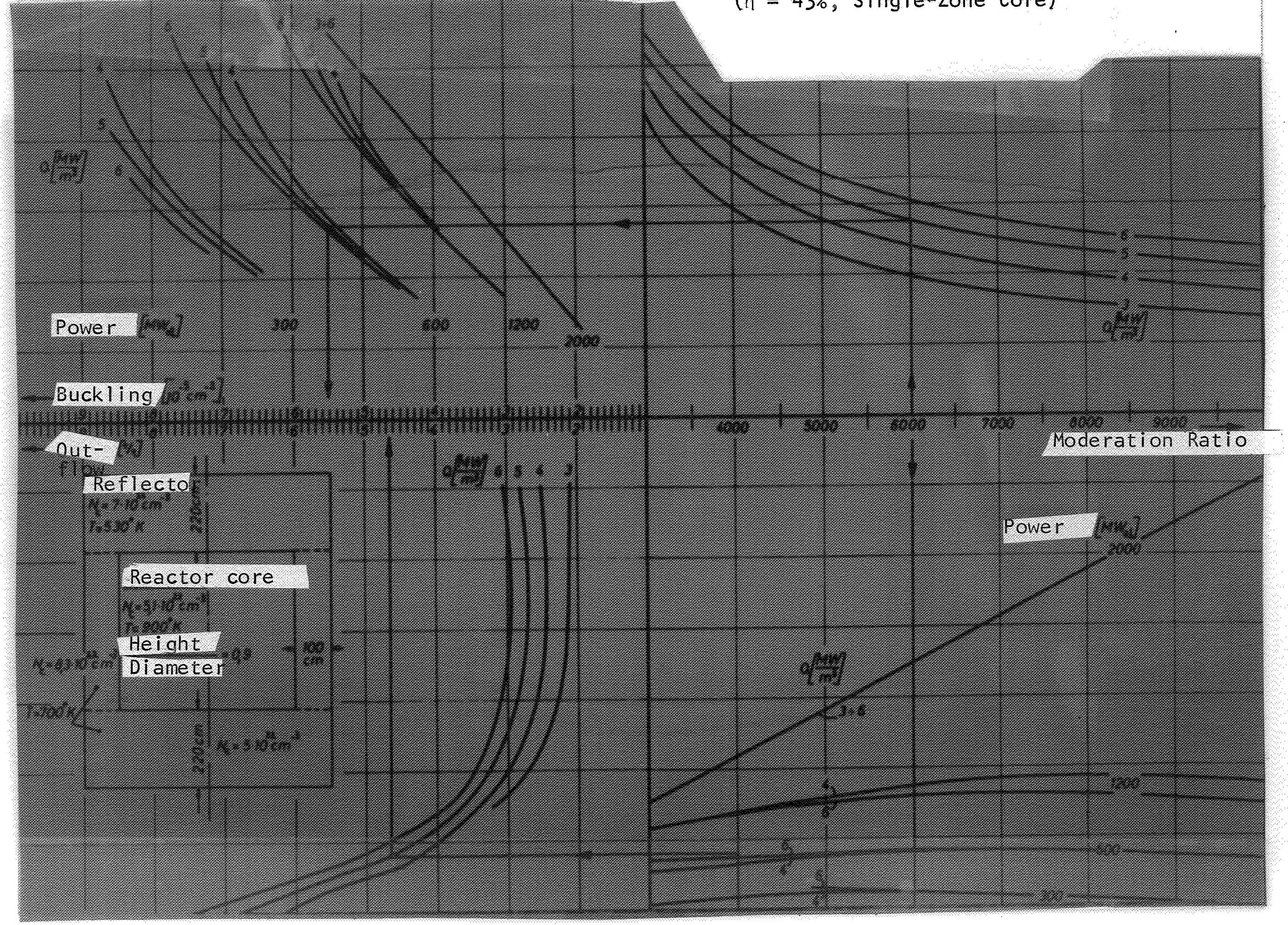


Diagram No. 2 Influence of the Cooling Time of the Fuel During Reprocessing on the Breeding Factor

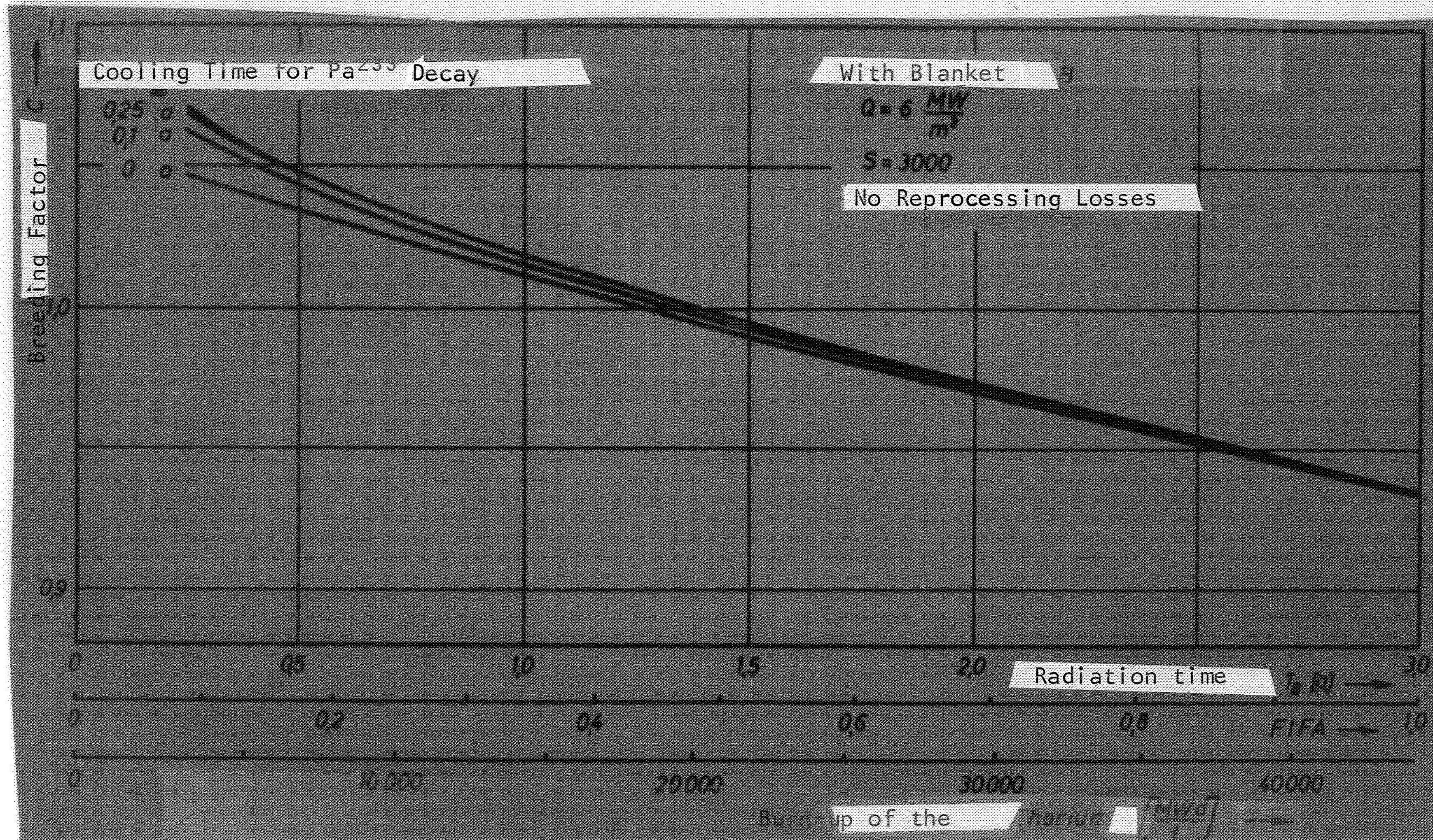


Diagram No. 3. Influence of the Cooling Time of the Fuel During Reprocessing

On Breeding Gain and Breeding Component

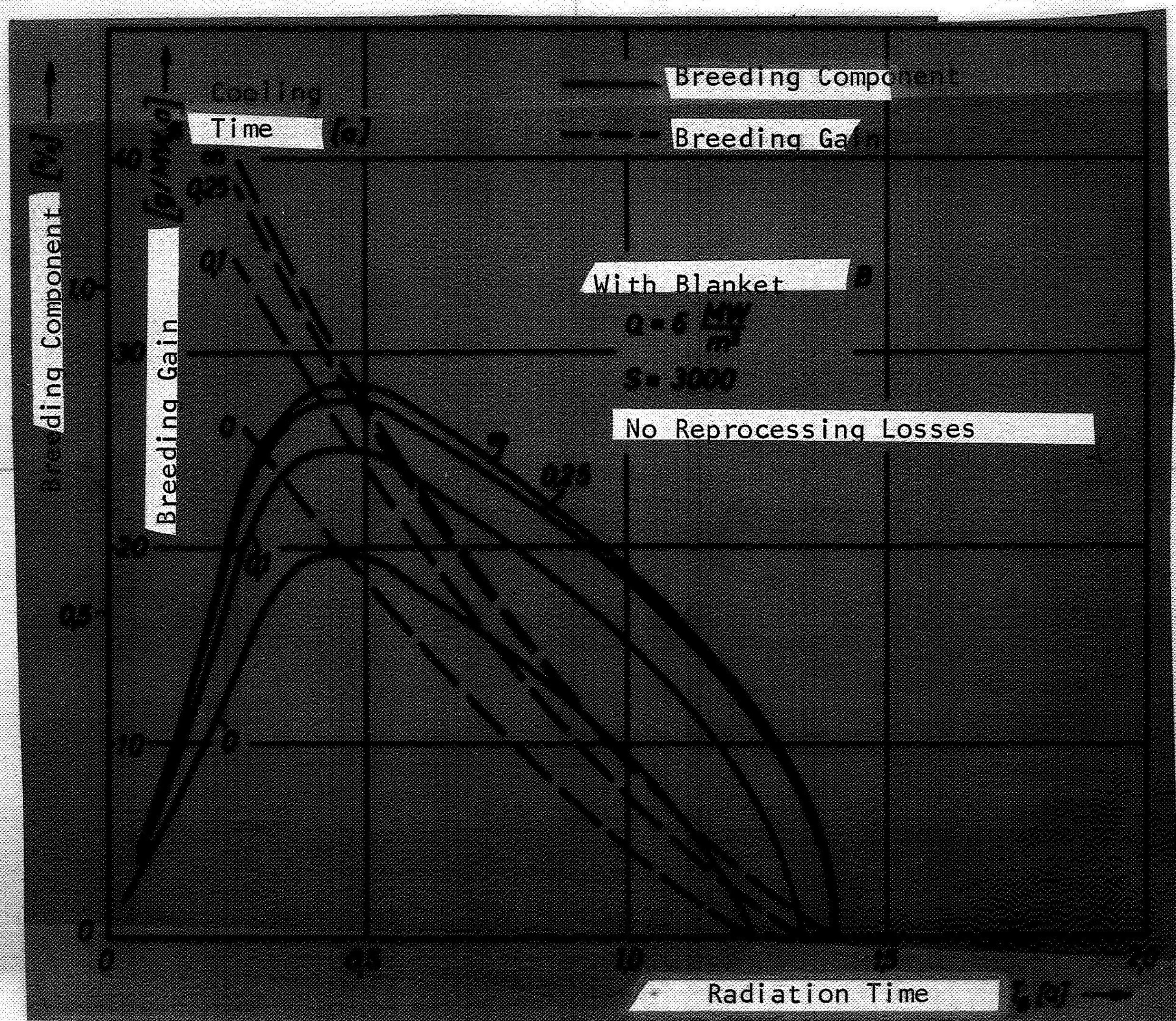


Diagram No. 4. Influence of the Reprocessing Losses on Breeding Gain.

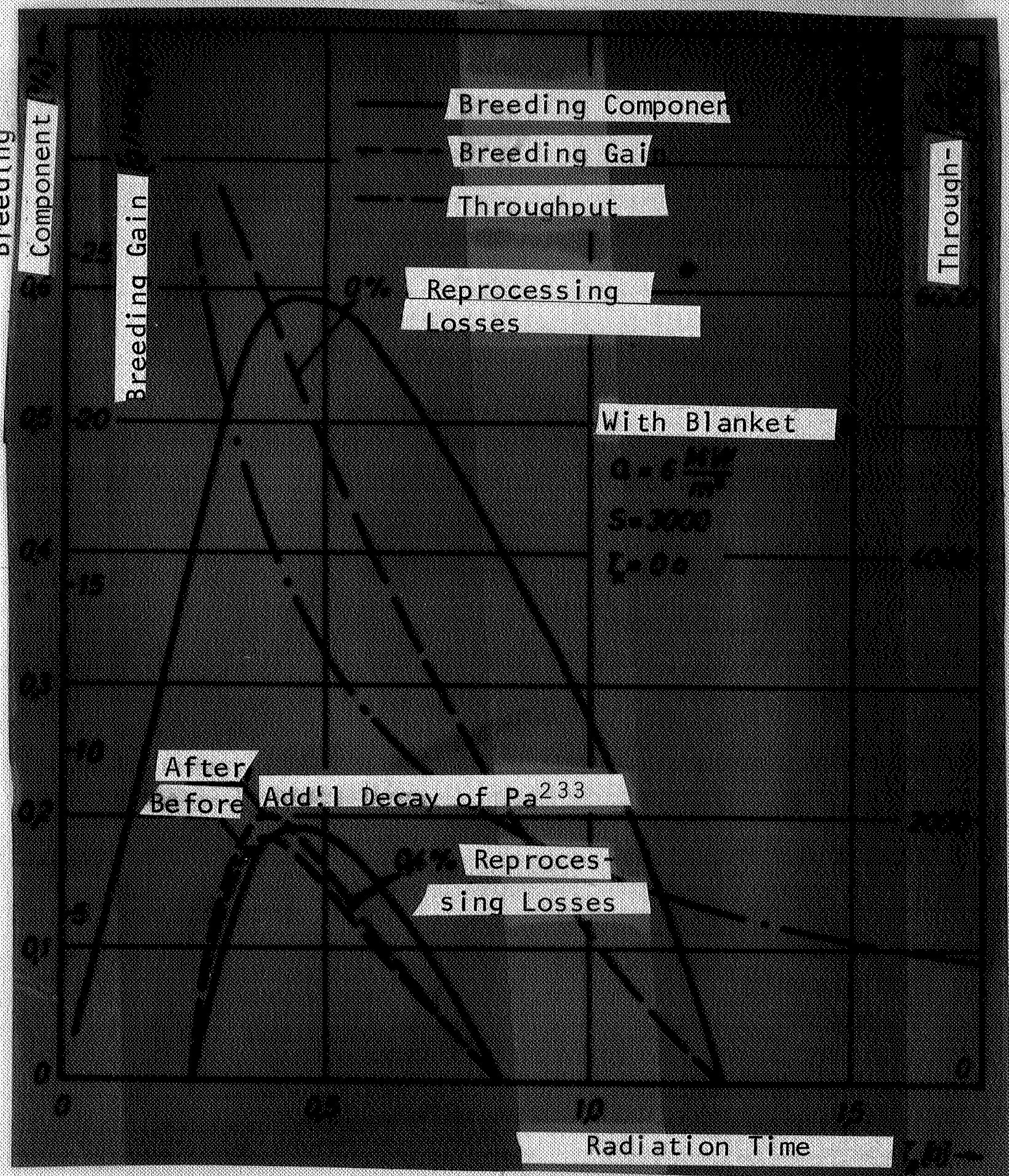
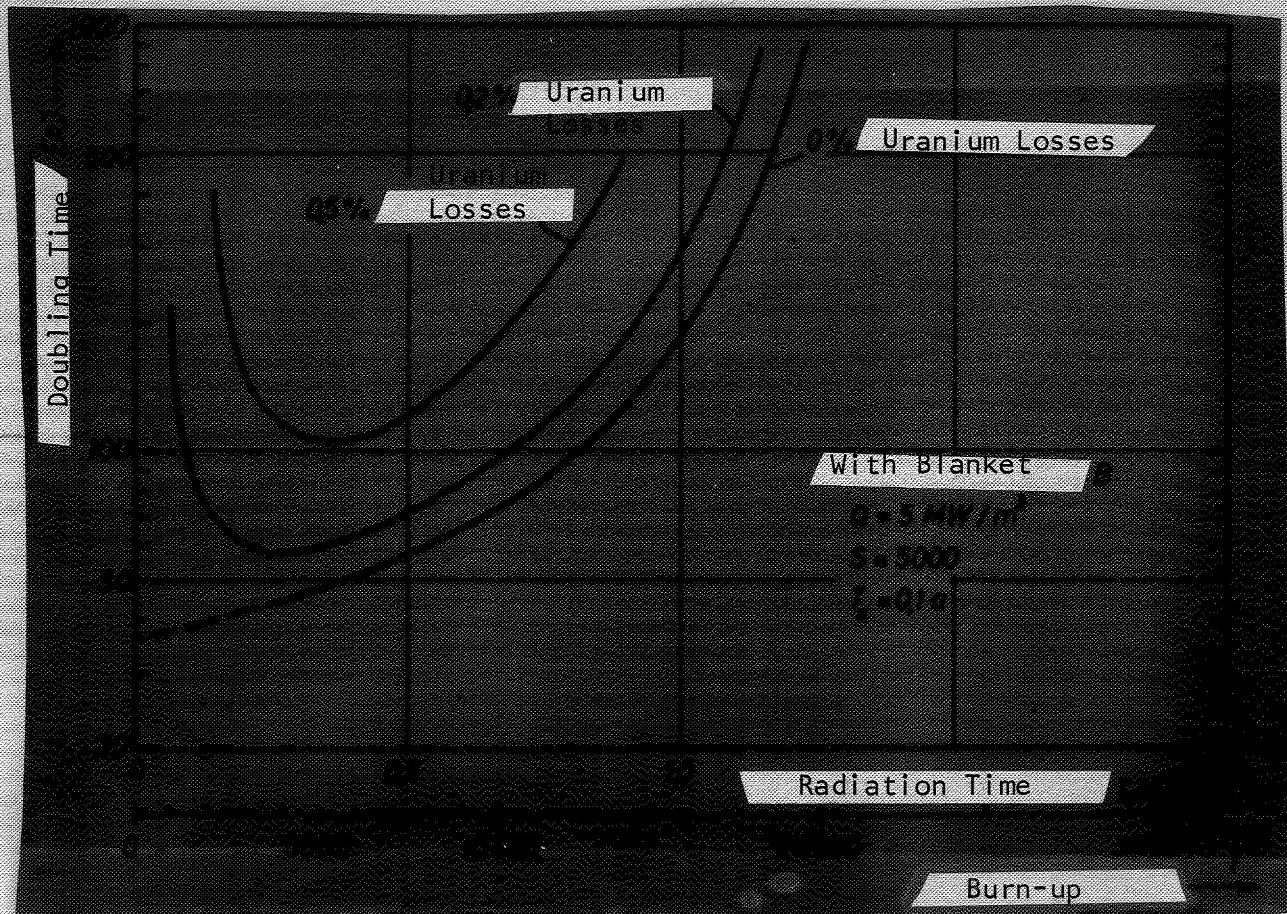


Diagram No. 5. Doubling Time as a Function of Burn-Up.



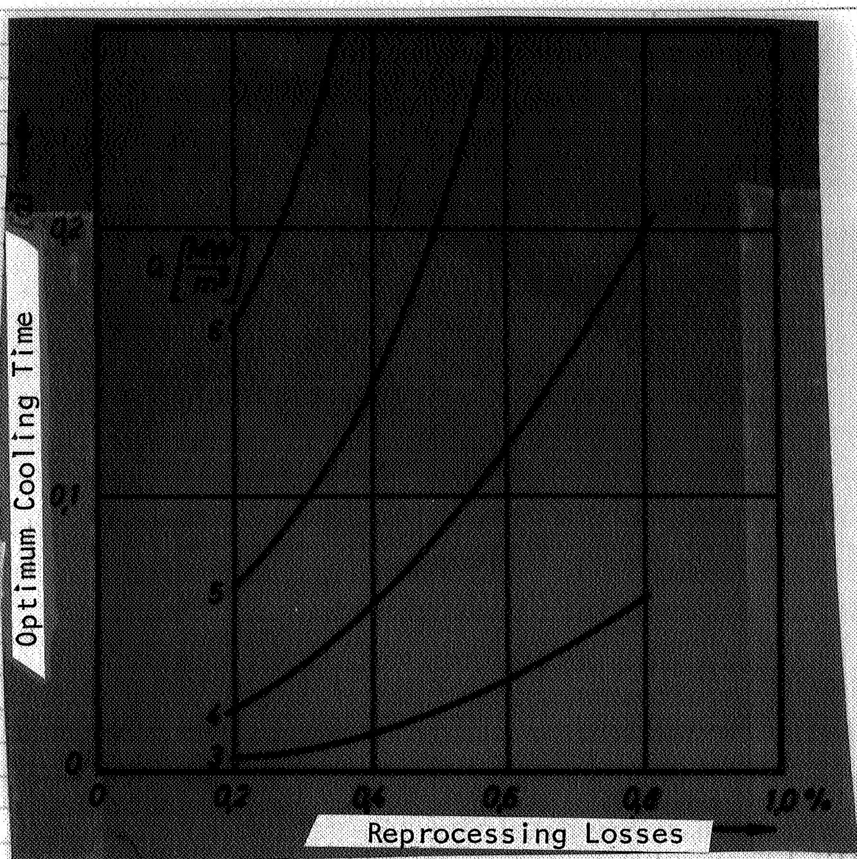


Diagram No. 6. Optimum Cooling Time of a Graphite-Moderated Thorium Breeder With Blanket A (2,000 MW_{el}).

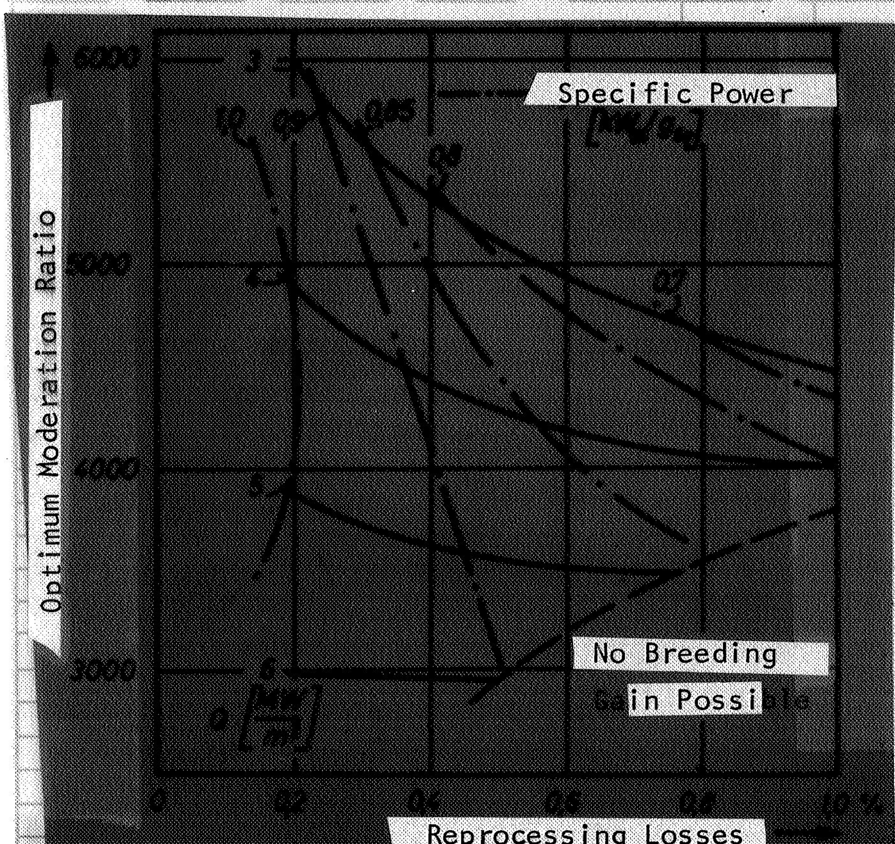


Diagram No. 7. Optimum Moderation Ratio of a Graphite-Moderated Thorium Breeder With Blanket A.

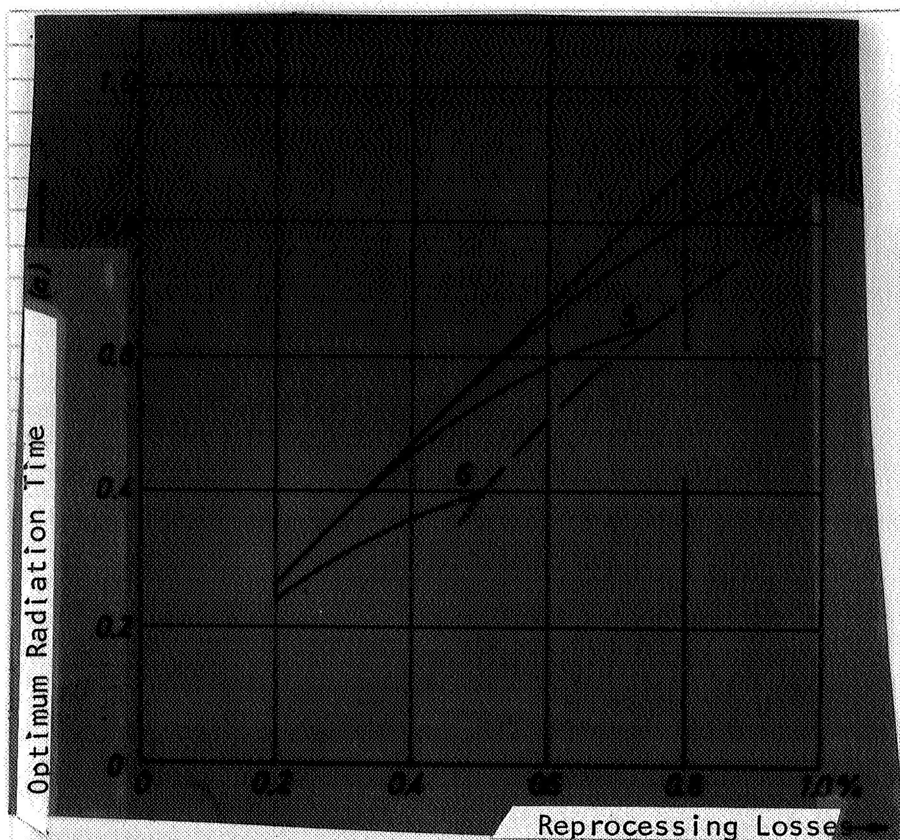


Diagram No. 8. Optimum Radiation Time of the Fuel Elements of a Thorium Breeder With Blanket A.

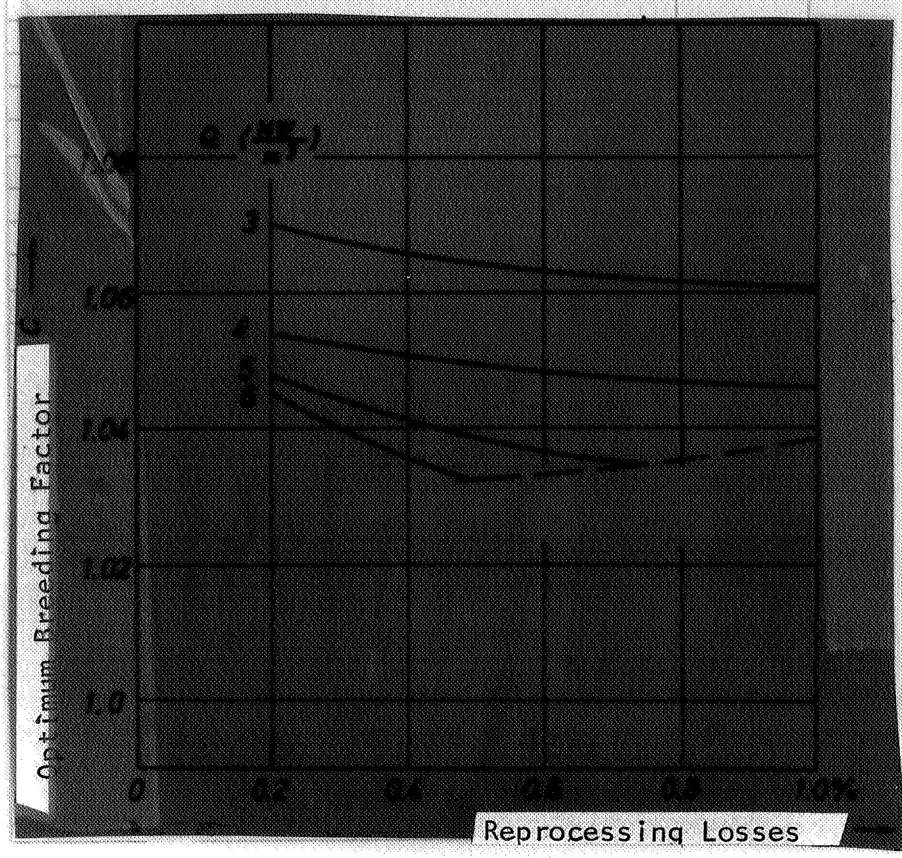


Diagram No. 9. Optimum Breeding Factor of a Thorium Breeder With Blanket A.

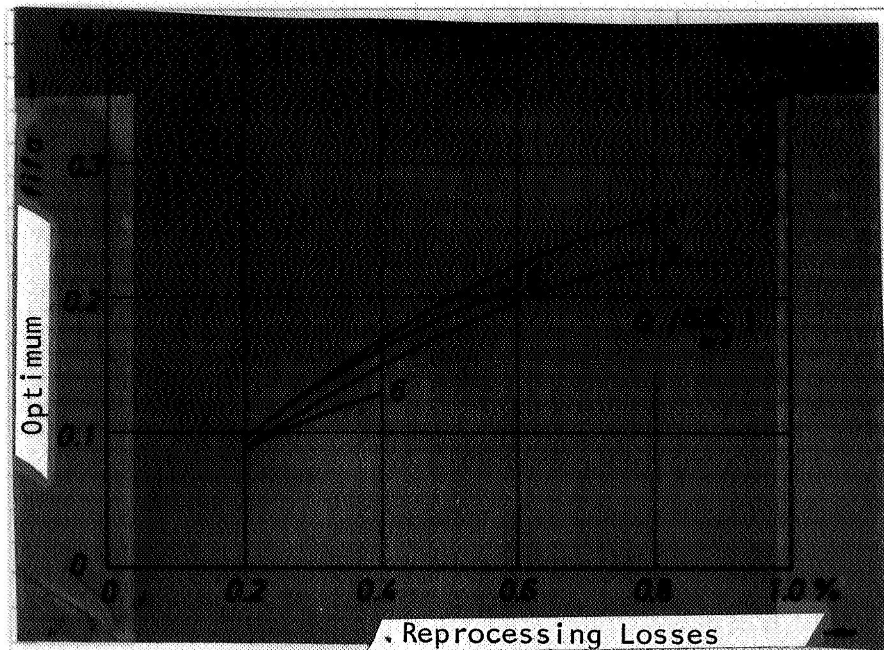


Diagram No. 10. Optimum FIFA For a Graphite-Moderated Thorium Breeder With Blanket A.

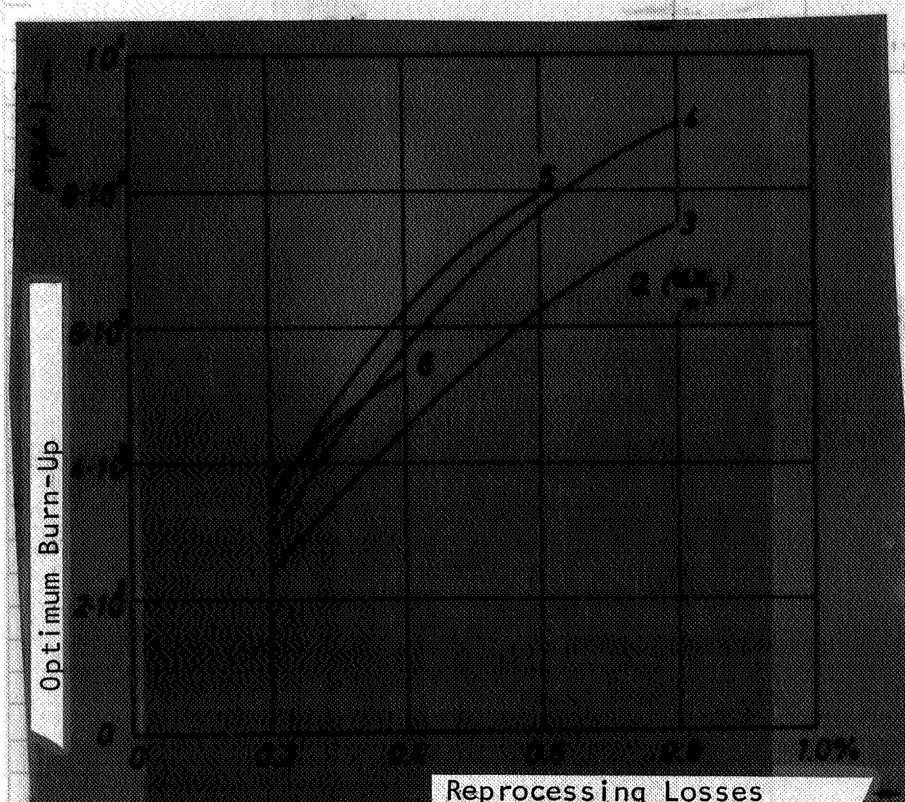


Diagram No. 11. Optimum Burn-Up of a Graphite-Moderated Thorium Breeder With Blanket A.

Diagram No. 12. Minimum Doubling Times of a Graphite-Moderated Thorium Breeder With Blanket A (2,000 MW_{el}).

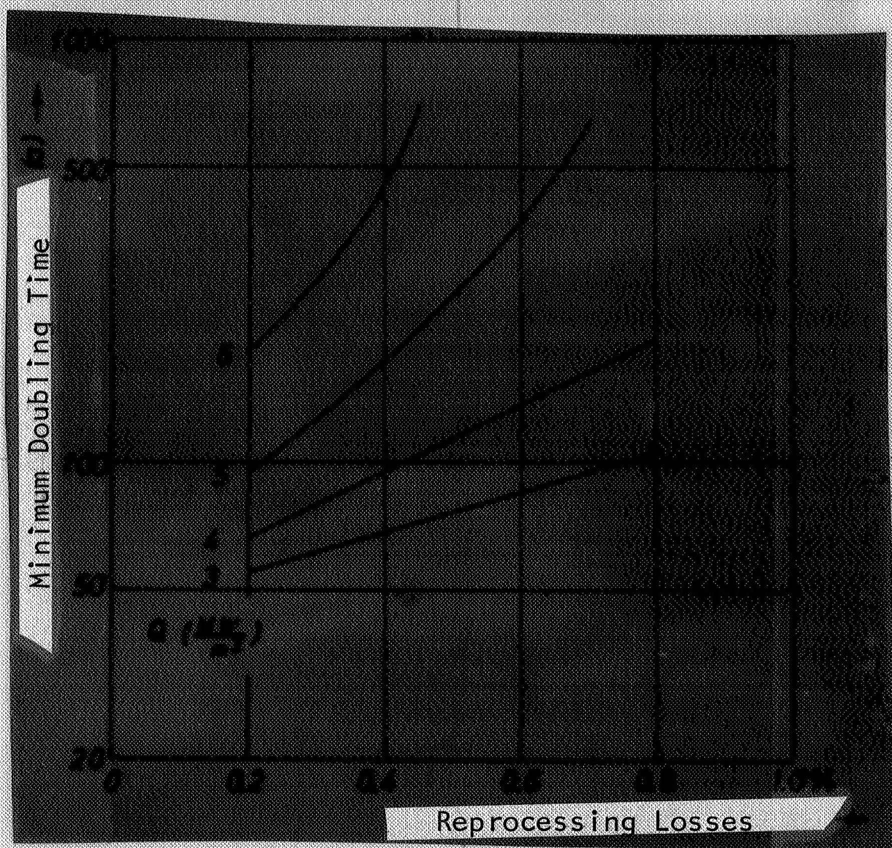
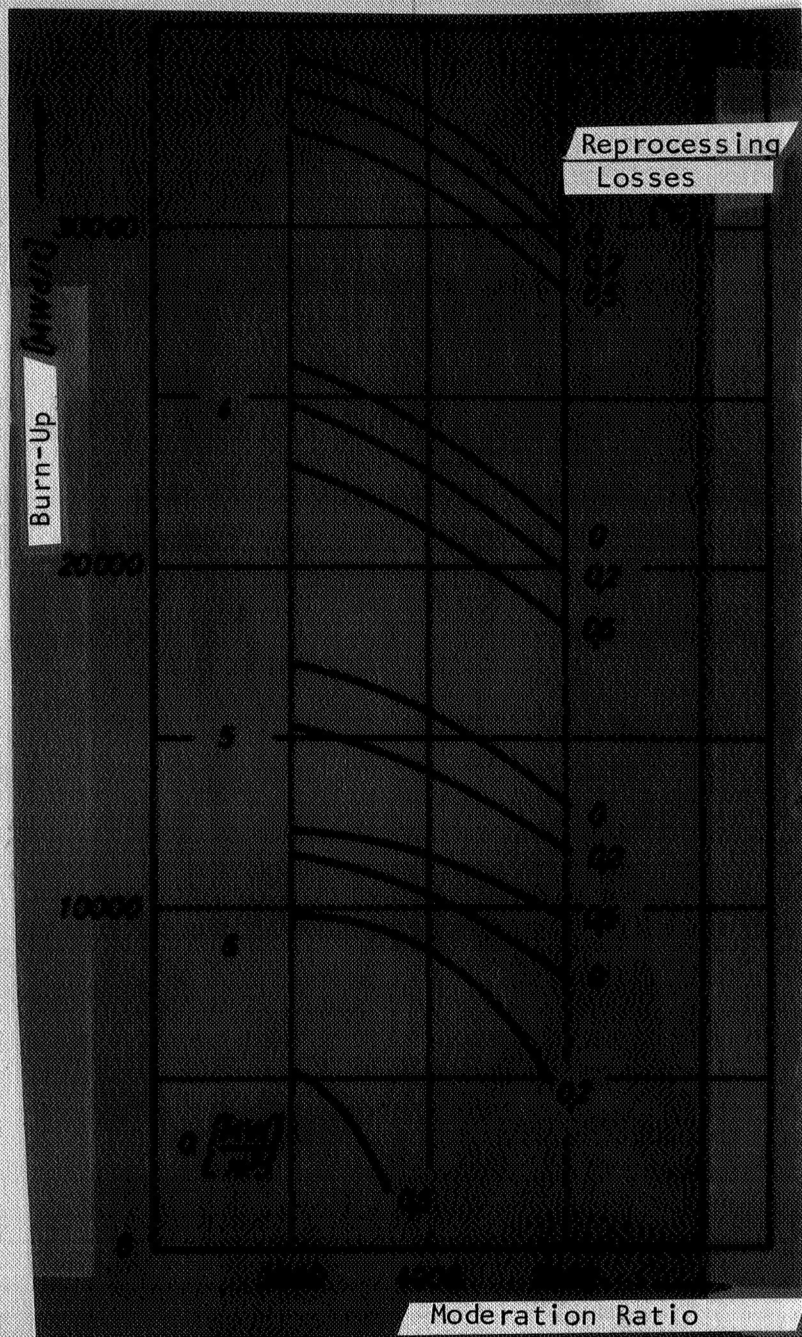


Diagram No. 13. Maximum Burn-Up at Infinite Doubling Time For a Graphite Moderated Thorium Breeder With Blanket A.



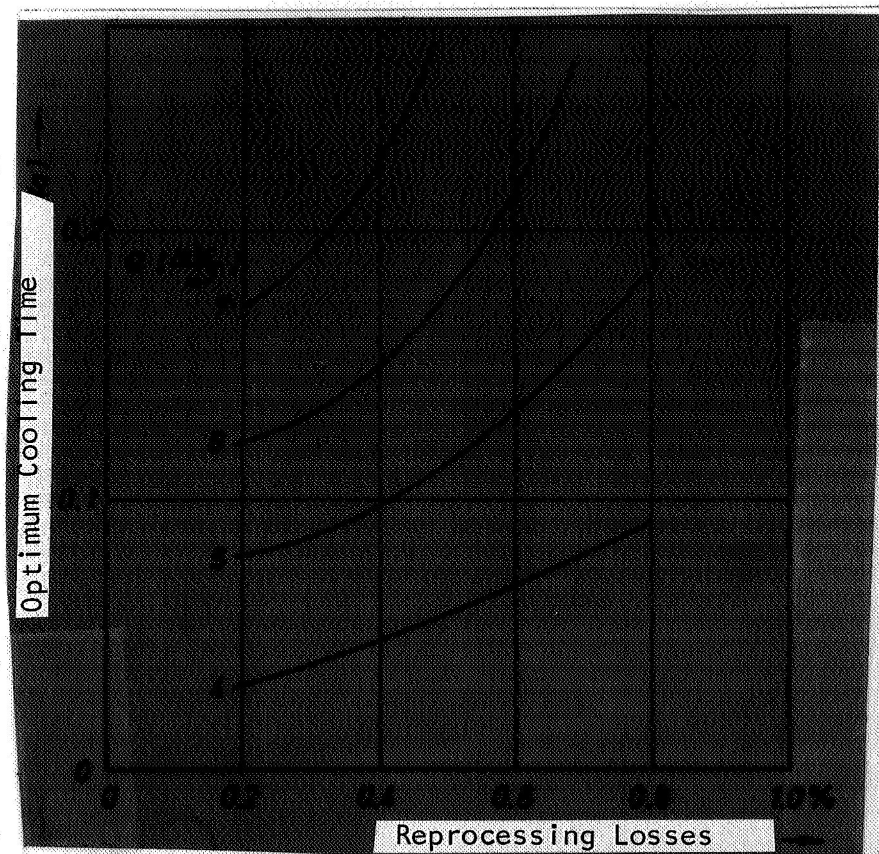


Diagram No. 14. Optimum Cooling Time of a Graphite-Moderated Thorium Breeder With Blanket B.

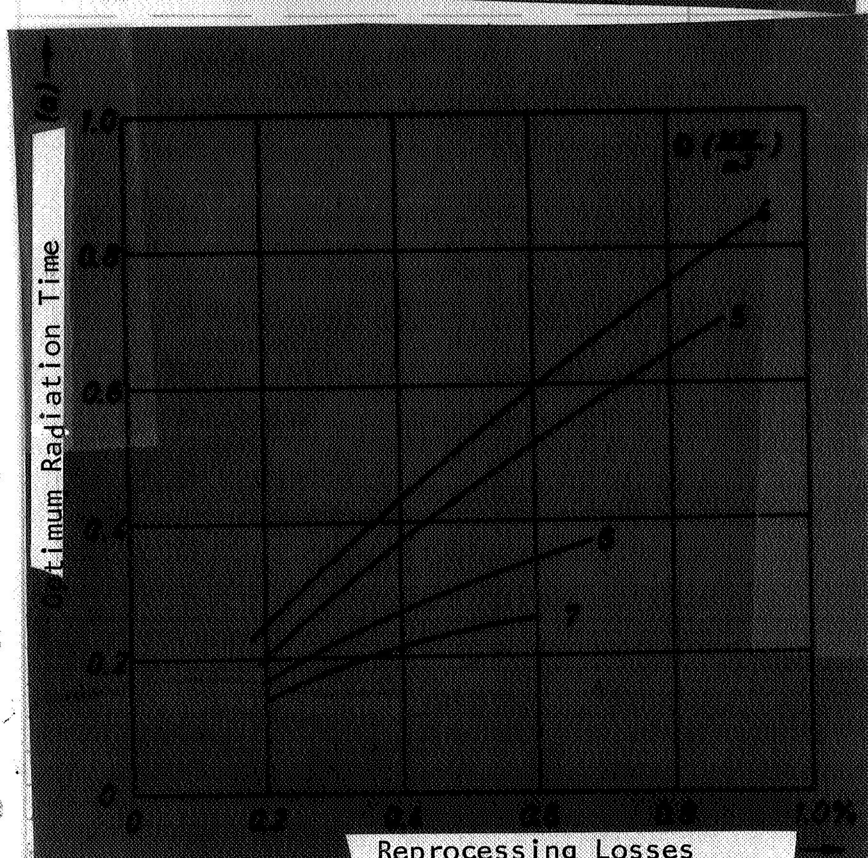
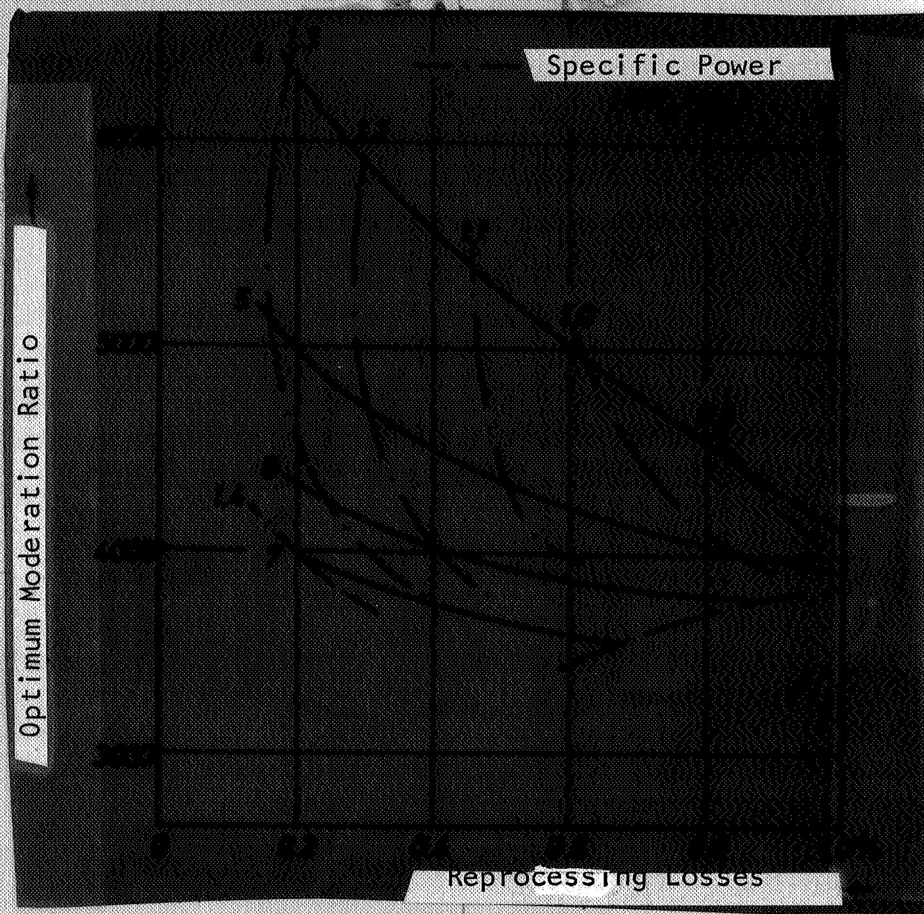


Diagram No. 15. Optimum Radiation Time of a Thorium Breeder With Blanket B.

Diagram No. 16. Optimum Moderation Ratio For a Graphite-Moderated Thorium Breeder With Blanket B.



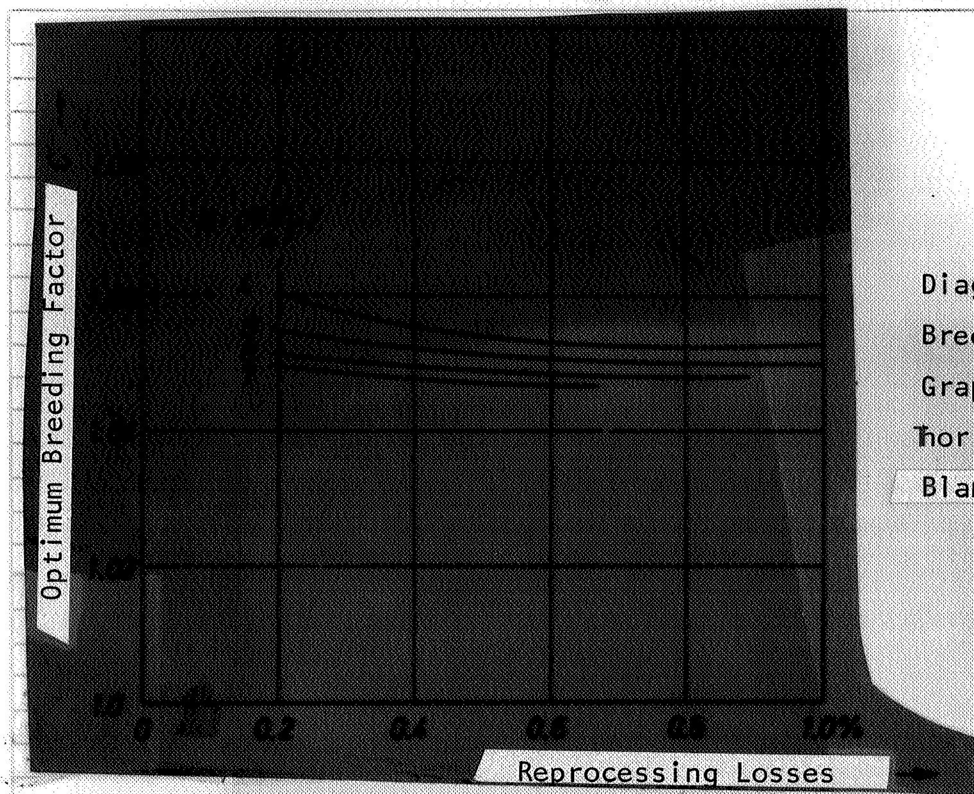


Diagram No. 17. Optimum Breeding Factor of a Graphite-Moderated Thorium Breeder With Blanket B.

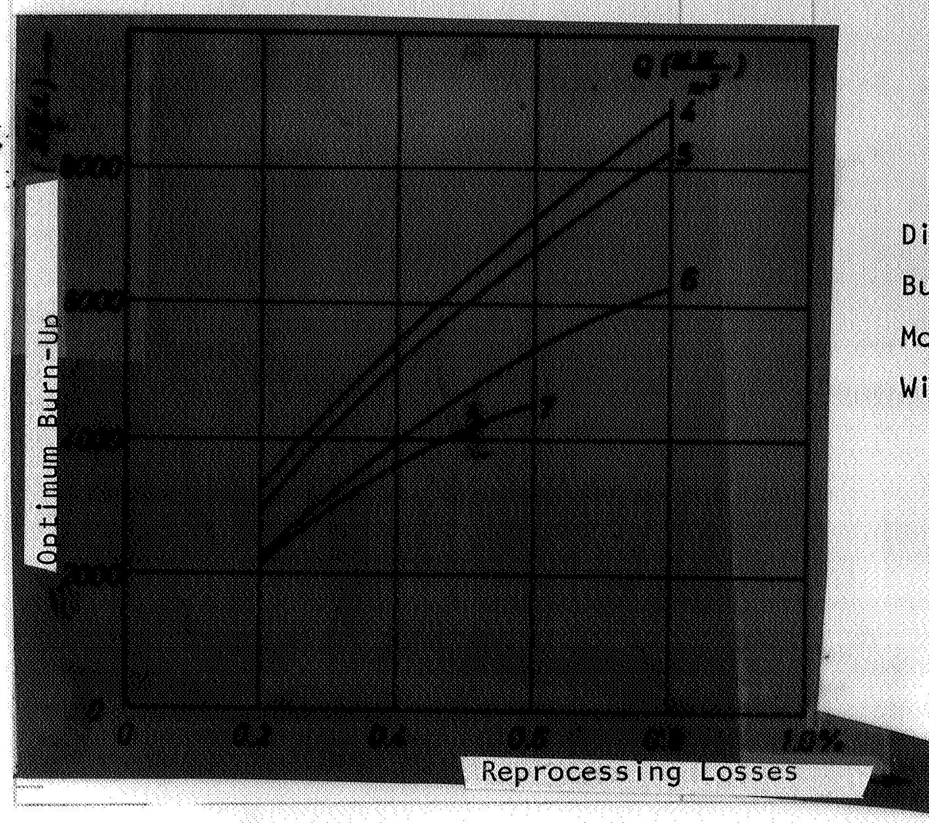
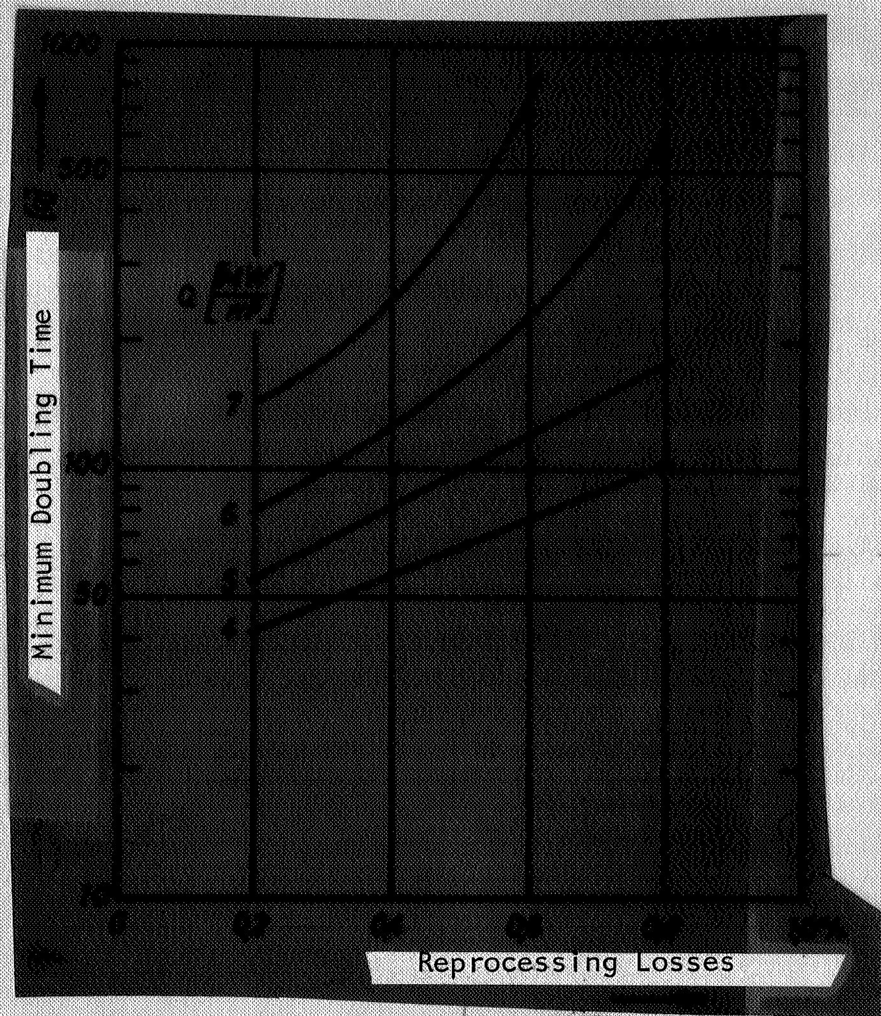


Diagram No. 18. Optimum Burn-Up Of a Graphite-Moderated Thorium Breeder With Blanket B.

Diagram No. 19. Minimum Doubling Times of a Graphite-Moderated Thorium Breeder With Blanket B (2,000 MW_{el})



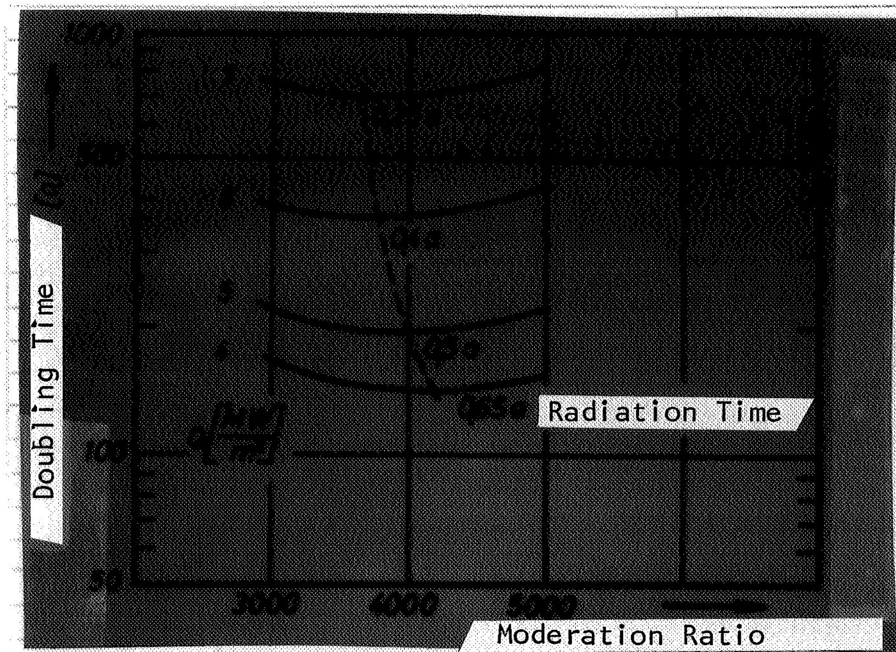


Diagram No. 20.
Doubling Times of a Thorium Breeder With Blanket B in Water Reprocessing.

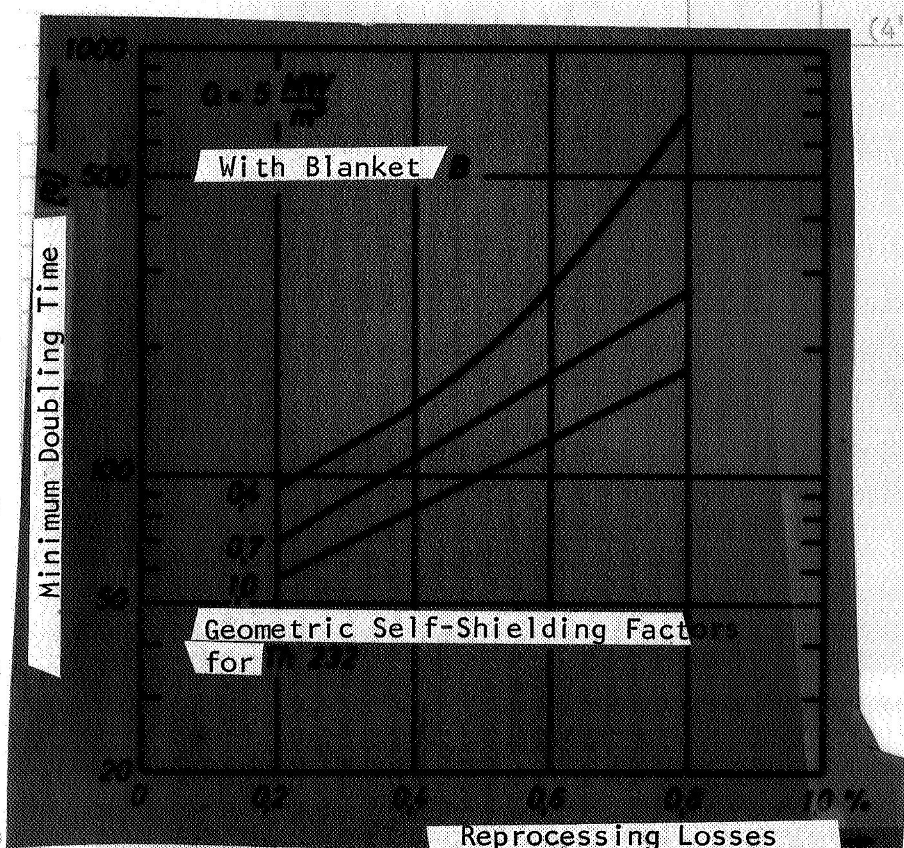


Diagram No. 21. Influence of the Self-Shielding Factors on the Doubling Time.

Diagram No. 22. Maximum Burn-Up at Infinite Doubling Time For a Thorium Breeder With Blanket B (2,000 MW_{el}).

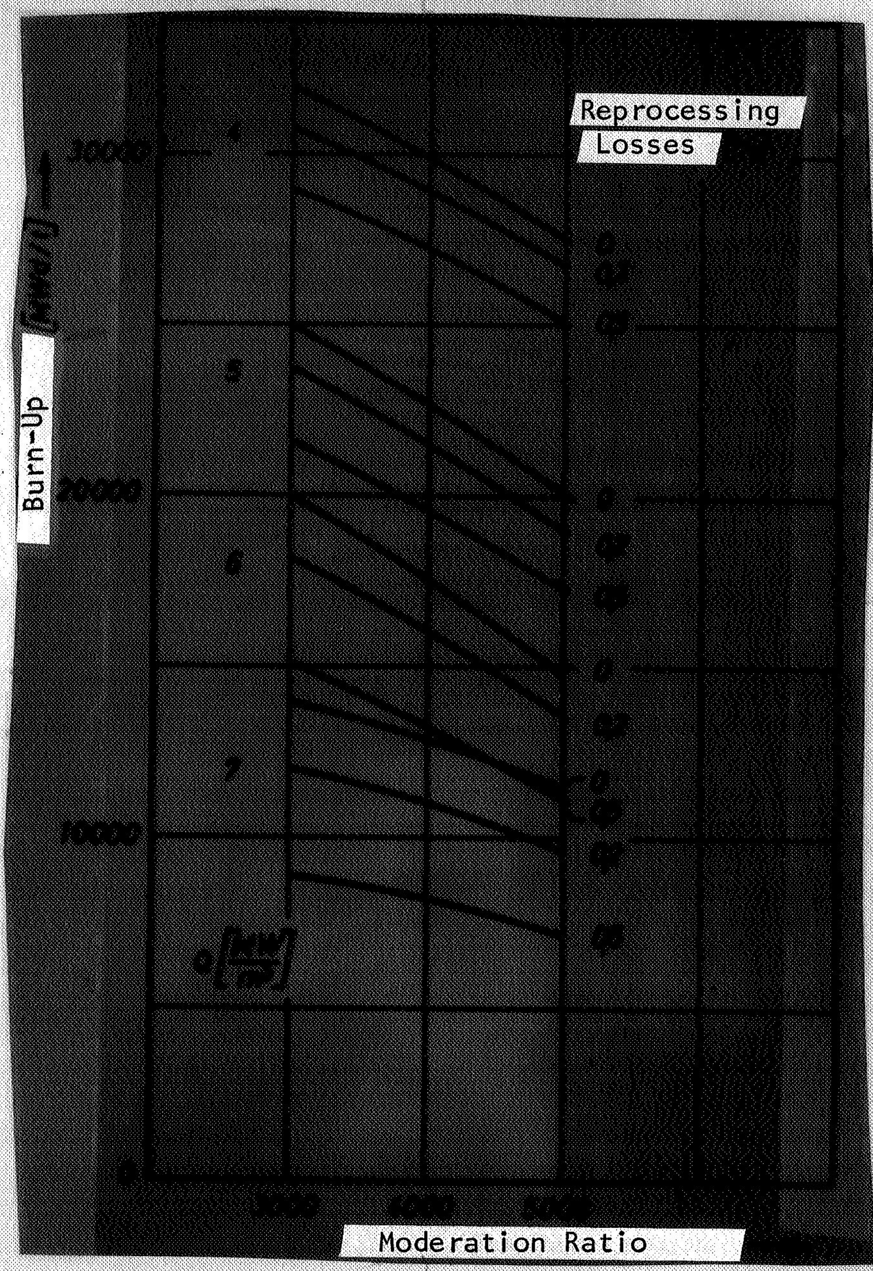


Diagram No. 23. Influence of Regeneration of the Fuel on the Duplication Time of a Graphite Moderated Thorium Breeder With Blanket B.

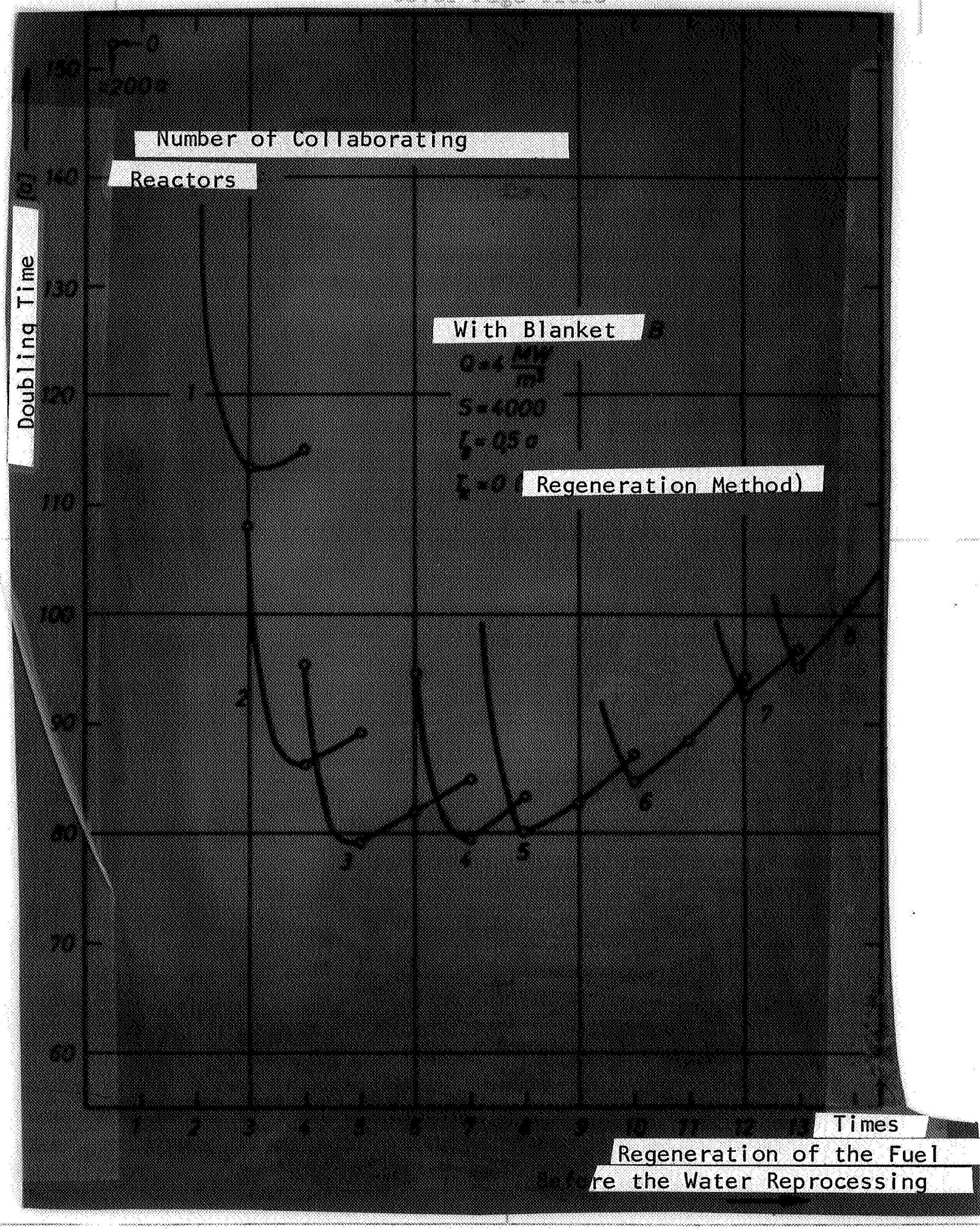


Diagram No. 24. Minimum Doubling Times of a Graphite-Moderated Thorium Breeder With Blanket B With the Use of Various Reprocessing Methods (Reactor Power 2,000 MW_{el}).